Tennessee Valley Authority, Post Office Box 2000, Soddy Daisy, Tennessee 37384-2000

January 18, 2011

10 CFR 50.73

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Unit 1
Facility Operating License No. DPR-77
NRC Docket No. 50-327

Subject: **Licensee Event Report 327/2010-002, Revision 0**

The enclosed Licensee Event Report (LER) provides details concerning a manual reactor trip and automatic engineered safety feature actuation of auxiliary feedwater following the failure of the steam generator (SG) level control to maintain SG levels.

The Tennessee Valley Authority is submitting this report in accordance with 10 CFR 50.73 (a)(2)(iv)(A), a condition that resulted in automatic actuation of the reactor protection system.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact B. A. Wetzel, SQN Site Licensing Manager, at (423) 843-7170.

Respectfully,

M Skaggs

Michael D. Skaggs
Site Vice President
Sequoyah Nuclear Plant

Enclosure: Licensee Event Report - Manual Reactor Trip as a Result of a Failure of Feedwater Control to Maintain Steam Generator Levels

cc: NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

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ENCLOSURE

LICENSEE EVENT REPORT 327/2010-002, MANUAL REACTOR TRIP AS A RESULT OF A FAILURE OF FEEDWATER CONTROL TO MAINTAIN STEAM GENERATOR LEVELS

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 10/31/2013
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Service Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Sequoyah Nuclear Plant (SQN), Unit 1	05000327	1 OF 6

4. TITLE: Manual Reactor Trip as a Result of a Failure of Feedwater Control to Maintain Steam Generator Levels

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
11	16	2010	2010	- 002 -	00	01	18	2011	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
26	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
SQN - Scott Bowman	(423) 843-6910

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE) ☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On November 16, 2010, at 2210 Eastern Standard Time, SQN Unit 1 was manually tripped based on decreasing steam generator (SG) level in Loop 4. Prior to the reactor trip, reactor power was at 26 percent and ascending following completion of a scheduled refueling outage. A manual turbine trip was initiated, because a moisture separator reheater safety relief valve lifted and would not reseat. Following the manual turbine trip, the Loop 4 SG feedwater level controller was unable to control the feedwater level because of an inoperable main steam dump valve controller. Manual actions were taken by Operations personnel, but level could not be recovered and the reactor was tripped. The root cause was determined to be a failure to identify and perform adequate installation testing on the main steam dump valve controller. During the refueling outage that ended on 11/15/2010, a digital feedwater design change was implemented requiring relocation of the main steam dump valve controller. Relocating the controller required the power cord to be unplugged, the controller moved, and the power cord to be reinstalled. There was no installation testing identified for this controller in the design change notice or the associated work order. When the power cord was reinstalled, it was not properly latched. This resulted in the controller becoming unplugged. An immediate corrective action required components that were moved during the modification to be inspected. Corrective actions include revising the modification installation and installation testing procedures to add additional verifications and reviews.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

I. PLANT CONDITION(S)

SQN, Unit 1 was operating at approximately 26 percent power during power ascension following the Unit 1 Cycle 17 refueling outage that ended 11/15/2010 when the manual reactor trip was initiated.

II. DESCRIPTION OF EVENT

A. Event:

On November 16, 2010, at 2210 Eastern Standard Time (EST), SQN, Unit 1 was manually tripped based on decreasing steam generator (SG) [EIIS Code AB] level in Loop 4. Prior to the reactor trip, reactor power was at 26 percent and ascending following completion of the Unit 1 Cycle 17 refueling outage. At 2148, the main control room (MCR) was notified that a moisture separator reheater (MSR) safety valve [EIIS Code RV] had lifted and would not reseat. At 2206, the turbine was manually tripped to isolate the steam leak. Following the turbine trip, automatic SG level control [EIIS Code JB] did not maintain SG level. Operations personnel took manual control; however, SG level could not be recovered. A manual reactor trip was initiated by Operations personnel as a result of a low narrow range SG level.

The feedwater level controller for the Loop 4 SG was unable to control the level because of the failure of a main steam dump valve controller [EIIS Code SB]. During a subsequent investigation, the main steam dump valve controller was found with the power plug partially disengaged.

During the refueling outage, the controller had been moved from one rack location to another to support the digital feedwater modification on SQN, Unit 1. This required the power cord to the controller to be unplugged, moved, and cables relocated. Upon reconnection of power, the power cord was not correctly plugged back into the controller. The controller function was not modified. There was no installation testing identified in the design change notice or the associated work order for the functionality of the controller.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

Main Steam Dump Valve Controller

C. Dates and Approximate Times of Major Occurrences:

Date	Description
November 16, 2010 at 2148 EST	The MCR was notified that a MSR safety relief valve had lifted. Operators entered Abnormal Operating Procedure (AOP)-S.05, "Steam or Feedwater Leak."
2202 EST	Operators entered AOP-C.03, "Rapid Shutdown or Load Reduction," as a result of being unable to isolate the steam leak.
2206 EST	Operators initiated a manual turbine trip. Following the turbine trip, the Loop 4 SG level controller did not automatically control level. Operators took manual actions but SG level could not be recovered.
2210 EST	Operators entered Emergency Procedure (E)-0, "Reactor Trip or Safety Injection," and initiated a manual reactor trip.
2236 EST	Operators exited applicable emergency procedures, and entered General Operating (GO) Instructions 0-GO-6, "Power Reduction From 30 Percent Reactor Power to Hot Standby."
November 19, 2010	An inspection performed under a work order identified that the power plug for the main steam dump valve controller was partially disconnected.

D. Other Systems or Secondary Functions Affected:

No other systems or secondary functions were affected by this event.

E. Method of Discovery:

The MCR was notified that a MSR safety relief valve had lifted.

F. Operator Actions:

Upon notification that the MSR safety relief valve had lifted, Operations took actions as required by plant procedures and performed a manual turbine trip. Following the turbine trip, automatic SG level control did not maintain SG level. Operations took manual control; however, SG level could not be recovered. A manual reactor trip was initiated by the operating crew. After the reactor trip, the operating crew took actions necessary to stabilize the unit in a safe condition and maintained the unit in hot standby, Mode 3.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

G. Safety System Responses:

The plant responded as expected for the conditions of the reactor trip.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The cause of initiating a manual turbine trip was a MSR relief valve lifted because of foreign material lodged between the seat and the disk of the gland sealing steam check valve. The immediate cause of the reactor trip was that following the turbine trip the main steam dump valve controller did not operate to control steam flows. This led to a manual reactor trip when the Loop 4 SG level fell below the time delayed reactor trip setting.

B. Root Cause:

The root cause of the reactor trip was determined to be a failure to identify and perform adequate installation testing on a main steam dump valve controller following its relocation as part of the digital feedwater modification performed during the Unit 1 Cycle 17 refueling outage. The failure to perform installation testing prevented the identification of the failure to properly connect the power cord to the steam dump controller resulting in a failure of the controller to perform its design function following the turbine trip.

C. Contributing Factor:

A contributing cause for this event was the failure to ensure that both of the locking devices engaged on the power cord when it was reattached. A thorough inspection would have identified that the power cable was not fully locked into place.

IV. ANALYSIS OF THE EVENT

SQN, Unit 1 reactor power was at 26 percent and ascending following completion of a scheduled refueling outage. Prior to the event, the reactor coolant system (RCS) pressure was approximately 2235 pounds per square inch gauge (psig). Following the turbine trip, RCS pressure peaked at 2267 psig before oscillating between 2190 and 2240 psig because of SG level swings. Following the reactor trip, the minimum RCS pressure was approximately 2117 psig, well above the pressure that would have initiated a safety injection signal (1870 psig). The RCS minimum temperature following the trip was approximately 533 degrees Fahrenheit and remained within technical specification (TS) limits. The minimum pressurizer level following the trip was 16 percent. This plant response was expected because of the low initial power level and low decay heat as the

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

plant was in power ascension following a refueling outage. No TS limits were exceeded and the Updated Final Safety Analysis Report (UFSAR) analysis of this event remained bounding.

The UFSAR states that the plant should be able to withstand a turbine trip up to 50 percent power without requiring a reactor trip. During this event, a manual reactor trip was initiated by Operations personnel after manual actions were taken to recover SG level were unsuccessful.

V. ASSESSMENT OF SAFETY CONSEQUENCES

Based on the above "Analysis of The Event," this event did not adversely affect the health and safety of plant personnel or the general public.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

Corrective actions included performing an inspection of the main steam dump valve controller that identified the power cord was unplugged and plugging it back in. A visual inspection of the six control components that were moved as part of the digital feedwater modification was performed.

Testing was performed in the SQN simulator to verify that an unplugged main steam dump valve controller would cause the conditions and events that occurred prior to the manual reactor trip. The results confirmed that a partially disconnected power cord to the controller would have caused a loss of SG level control that necessitated the trip.

B. Corrective Actions to Prevent Recurrence:

The corrective actions are being managed through the SQN Corrective Action Program.

Applicable plant procedures regarding modification installation and installation testing will be revised to add additional verifications and reviews.

VII. ADDITIONAL INFORMATION

A. Failed Components:

None.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

B. Previous LERs on Similar Events:

A review of previous reportable events within the last three years identified that there has been a manual reactor trip following a turbine trip as a result of a MSR safety relief valve being unable to seat (LER 1-2009-004); however, there were no previous similar events that were a result of inadequate installation testing.

C. Additional Information:

None.

D. Safety System Functional Failure:

This event did not result in a safety system functional failure in accordance with 10 CFR 50.73(a)(2)(v).

E. Unplanned Scram with Complications:

This condition did not result in an unplanned scram with complications.

VIII. COMMITMENTS

None.